A Benchmark Experiment for Fast Neutron Transport in Graphite

I. 13 A Benchmark Experiment for Fast Neutron Transport in Graphite

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Present study has been motivated from material irradiation test and radiation shielding concerning with fusion reactor or fast neutron therapy.[1,3] Benchmark experiments for high energy neutron transport up to about 35 MeV in graphite have been performed at the T-O-F facility[4] using fast neutrons from thick beryllium target bombarded by 35 MeV pulsed proton beam. Neutrons are collimated through three kinds of shields; concrete and iron of total thickness of 1 m, iron of 30 cm thick and the third composed of concrete, borated paraffine, lead and iron. A neutron spectrometer of 5" × 2" NE-213 liquid scintillator was used. The detection efficiency versus neutron energy was determined from the measured response functions normalized with the Monte Carlo code O5S, which was previously reported.[5]

Three set of the neutron data have been obtained in the present experiment.

1) Transmission spectra of the neutrons were measured for the graphite slabs of thickness 10, 20 and 40 cm placed at 12.6 m from the neutron source. The neutron detector was located at 13.5 m from the neutron source. The T-O-F spectra were converted to neutron energy spectra. The results are shown in fig. 1 along with the values for carbon of the evaluated file ENDF/B=IV.

2) Scattering spectra from the graphite assembly of 30 cm cube were measured with the spectrometer at the angles of 45°, 90° and 135°. Neutron energy spectra shown in fig. 2 were obtained. In the analysis of the experiment, Monte Carlo calculations of the transport of neutrons were performed using the multi-group code MORSE.[6] The basic cross-section data set used for carbon was DLC-58[7] which was a multigroupe library consisted of 47 groups for neutron energies from thermal to 60 MeV on basis of the ENDF/B-IV for below 15 MeV and the theoretical model calculations for above 15 MeV. Comparisons of measured and calculated spectra are shown in fig. 3.

3) Energy spectra of secondary neutrons are also obtained with the unfolding of pulse height distribution from output of the NE-213 spectrometer. Figure 4 shows comparison of measured and calculated spectra at 90° as an example. Negative flux is caused by little terms of the Legendre polynomials taken on the calculation.

Many informaitons to the cross-section data set and the neutron transport calculation are obtained from the present benchmark experiment. The details will be published in near future.

References

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5) Takahashi H., Uchida H. and Sugiyama K., ibid. p. 113.
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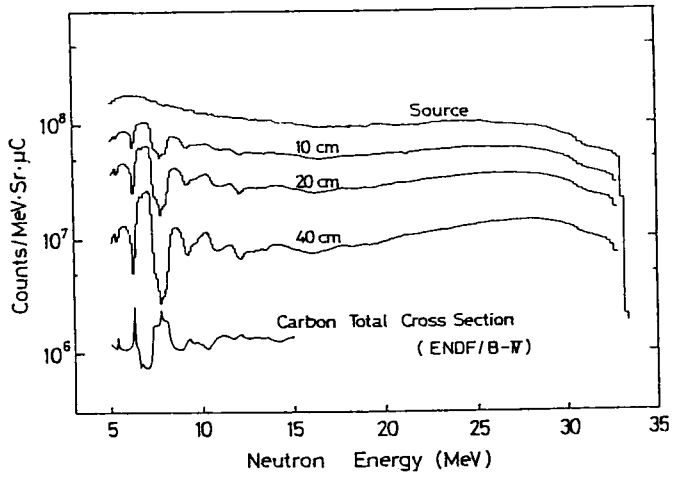

Fig. 1. Neutron transmission spectra through graphite. Neutron source spectrum from thick beryllium target bombarded by 35 MeV protons is also illustrated.

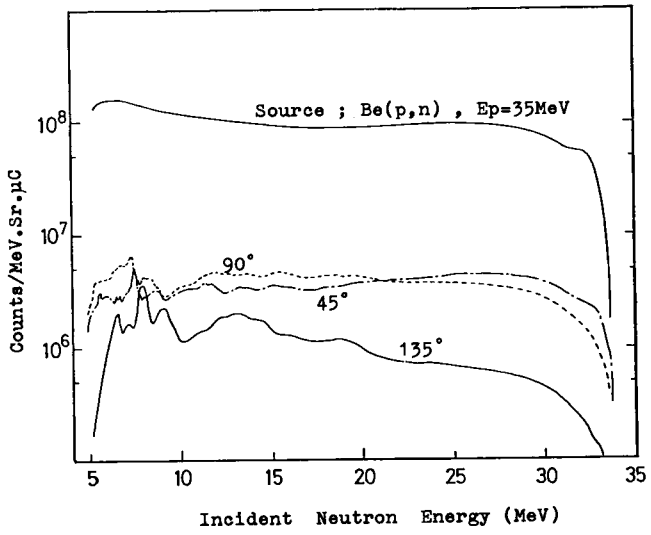

Fig. 2. Scattering neutron spectra from graphite assembly of 30 cm cube.

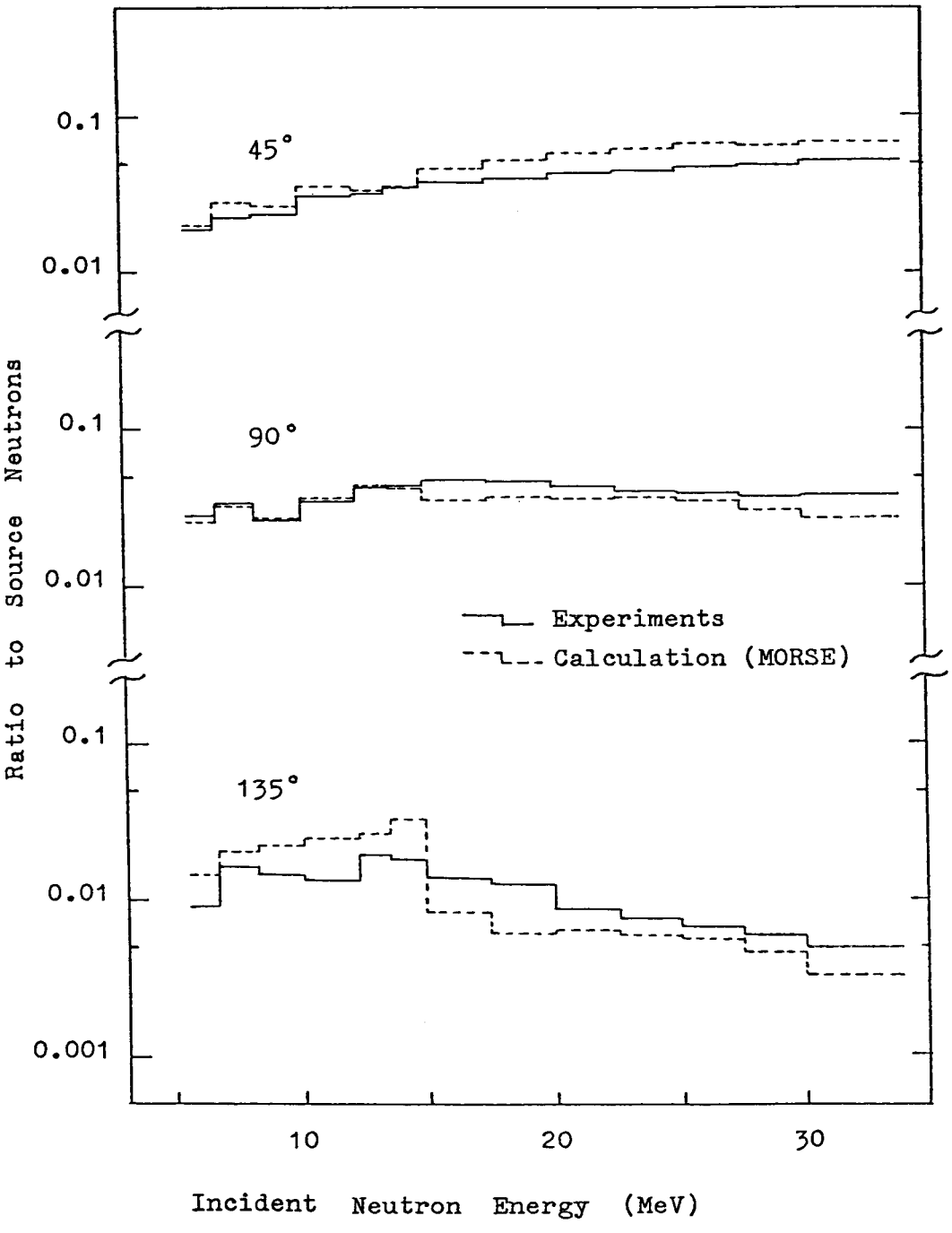

Fig. 3. Comparison of measured and calculated spectra of scattering neutrons from graphite assembly of 30 cm cube.

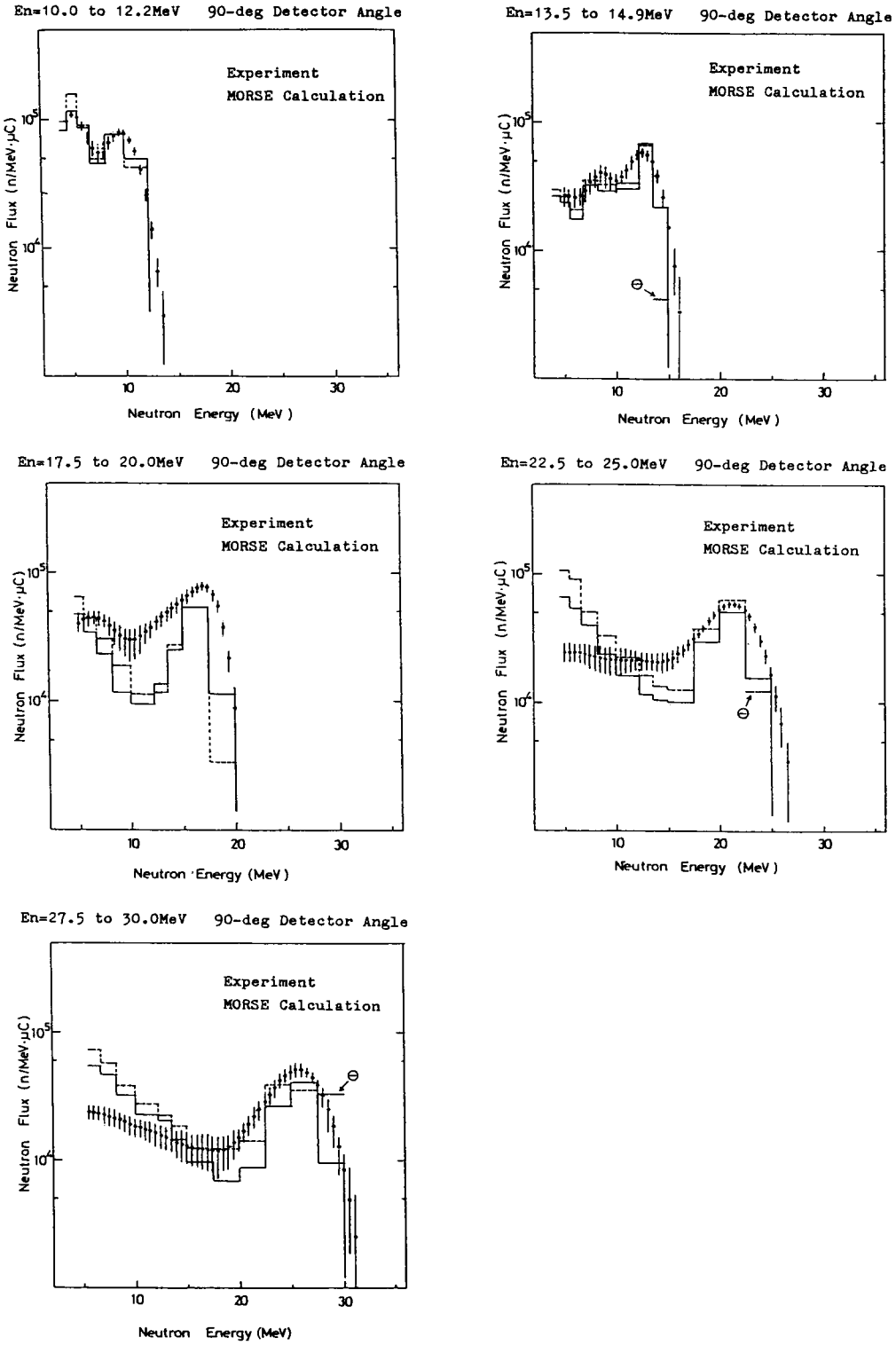

Fig. 4. Comparison of measured and calculated energy spectra of secondary neutron from the graphite assembly. Incident neutron energy is indicated by En.